ACCESSION #: 9909020208

NON-PUBLIC?: N

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Waterford Steam Electric Station, Unit 3 PAGE: 1 OF 9

DOCKET NUMBER: 05000382

TITLE: Reactor Shutdown Due to Loss of Controlled Bleed-Off Flow

EVENT DATE: 08/01/1999 LER #: 99-011-00 REPORT DATE: 08/31/1999

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

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Licensing Engineer

COMPONENT FAILURE DESCRIPTION:

CAUSE: X SYSTEM: RCS COMPONENT: AB-P MANUFACTURER: B580

REPORTABLE EPIX: Yes

SUPPLEMENTAL REPORT EXPECTED:

ABSTRACT:

On August 1, 1999, at 2136, with Waterford 3 operating at 100% power, an alarm for Middle Seal Pressure Low on Reactor Coolant Pump (RCP) 2B was received. Upon investigation, Operations personnel discovered lowering RCP seal pressures, along with dropping Controlled Bleed-Off (CBO) flow and increasing CBO temperature. Operations personnel entered the appropriate Off-Normal procedure for a reactor coolant pump malfunction. Initial actions were to attempt to lower CBO temperature by lowering Component Cooling Water (CCW) temperature to the RCP heat exchanger. Attempts to

lower CBO temperature were unsuccessful. Within 13 minutes, CBO temperatures increased to 204 degrees F, CBO flow dropped to zero gallons per minute, and middle and upper seal pressures dropped to 100 psi. Operations personnel manually tripped the reactor and secured RCP 2B. The cause of this event is believed to be fatigue-induced failure of the rotating baffle of RCP 2B. This event did not compromise the health and safety of the public.

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REPORTABLE OCCURRENCE

On August 1, 1999, at 2136 with Waterford 3 operating at 100% power, an alarm for Middle Seal Pressure Low on Reactor Coolant Pump (RCP) 2B was received. Upon investigation, Operations personnel discovered lowering RCP seal pressures, along with dropping Controlled Bleed-Off (CBO) flow and increasing CBO temperature. Operations personnel entered the appropriate Off-Normal procedure for a RCP malfunction. Initial actions were to attempt to lower CBO temperature by lowering Component Cooling Water (CCW) temperature to the RCP heat exchanger. Attempts to lower CBO temperature were unsuccessful. Within 13 minutes, CBO temperatures increased to 204 degrees F, CBO flow dropped to zero gallons per minute (gpm), and middle and upper seal pressures dropped to 100 psi. Operations personnel manually tripped the reactor and secured RCP 2B. This event is reportable under 10CFR50.73(a)(2)(iv) as an actuation of an Engineered Safety Feature or the Reactor Protection System.

INITIAL CONDITIONS

At the time of this event, Waterford 3 was operating in Mode 1 at 100% power. There was no major equipment out of service specific to this event and no TS Limiting Conditions for Operation Action Statements were in effect specific to this event.

EVENT DESCRIPTION

On August 1, 1999, at 2136 with Waterford 3 operating at 1 00% power, an alarm for Middle Seal Pressure Low on Reactor Coolant Pump (RCP) 2B [AB-P] was received. Upon investigation, Operations personnel discovered lowering RCP seal pressures, along with dropping Controlled Bleed- Off (CBO) flow and increasing CBO temperature. Operations personnel entered the appropriate Off- Normal procedure for a RCP malfunction. Initial actions were to attempt to lower CBO temperature by

lowering Component Cooling Water (CCW) [CC] temperature to the RCP heat exchanger [HX]. Attempts to lower CBO temperature were unsuccessful. Within 13 minutes, CBO temperatures increased to 204 degrees F, CBO flow dropped to zero gpm, and middle and upper seal pressures dropped to 100 psi. Operations personnel manually tripped the reactor and secured RCP 2B. The reactor coolant pumps are Byron Jackson vertically oriented single stage centrifugal pumps, Type 36x36x39 DFSS. These pumps have three face type mechanical seal stages in series with a fourth vapor stage to seal RCS pressure of 2250 psi. Pressure Breakdown Devices (PBD) [OR] (capillary tubes) are provided (one for each of the three face type mechanical seals). Each PBD carries a leakage flow in parallel with the face type mechanical seals of each stage. PBD's are designed to drop the pressure across each face seal such that full RCS pressure will not be exhibited to a single seal face during operation.

The mechanical seals [SEAL] are lubricated and cooled by a 1.5 gpm controlled reactor coolant leak-off. Reactor coolant enters the seal area at about 1.5 gpm from the heat exchanger/rotating baffle [BAF] area. RCS coolant flowing through the seal area is cooled by a 45-60 gpm flow of CCW supplied to the reactor coolant pump heat exchanger assembly. The RCP heat exchanger assembly contains passages for CCW to remove heat from the reactor coolant which lowers RCS temperature from approximately 550 degrees F to approximately 140 degrees F in the seal cavity.

RCS coolant enters below the heat exchanger near the pump shaft. Flow is directed up and around two heat exchanger cylinders by two cylinders of a rotating baffle. The rotating baffle is attached to the pump shaft by means of a bolted joint. The purpose of the two heat exchanger

cylinders and two rotating baffle cylinders is to provide more RCS surface area contact with the heat exchanger for cooling, and to ensure proper mixing to minimize thermal stratification. Initial disassembly involved removal of the sea[for RCP 2B and inspection of the rotating baffle. Inspection showed the first stage seal PBD tubing plugged with several small metallic slivers. Initial inspection also showed the baffle's bolted joint securely attached. However, the baffle had an observed through wall crack across the outer top surface continuing 180 degrees around the inner surface of the inner cylinder. Removal of the baffle showed the through-wall crack extended 160 degrees around and diagonally down the outer cylinder to within approximately 1 of the bottom. A visual inspection suggests the baffle crack was fatigue induced.

CAUSAL FACTORS

1. Design Configuration And Analysis

Inadequate Review of Design Change:

The pump Original Equipment Manufacturer (OEM) was contacted concerning the cracked rotating baffle. Discussions determined that this baffle design had not been analyzed for cyclic loads since the baffle was considered a low-stress pump component. The baffle configuration had been changed from a two-piece bolted arrangement to a one-piece arrangement. During the change to the one-piece arrangement the wall thickness of the upper inner cylinder was decreased from 11/16 inches to 5/16 inches. Additionally, the manufacturing process for the one-piece baffle had been changed from casting to forging.

2. Equipment Specification, Manufacture, And Construction

Manufacturing Process Which Could Have Introduced a Material Flaw In Baffle:

Visual inspection after failure suggested fatigue cracking. A material or manufacturing induced flaw could be the origin of the fatigue crack. Non-destructive examination (NDE) requirements imposed during the manufacturing of the failed baffle would not detect a sub-surface flaw in the material.

CORRECTIVE ACTIONS

1. Design Configuration and Analysis

Inadequate Review of Design Change

Perform an analysis of all forces acting on the one- and two-piece rotating baffles utilizing a Finite Element Model.

2. Equipment Specification, Manufacture, and Construction

Manufacturing Process Which Could Have Introduced a Material Flaw In Baffle

Perform a failure analysis of the cracked rotating baffle and associated debris.

SAFETY SIGNIFICANCE

The actual safety significance of this event was negligible. Due to prompt operator action to trip the reactor and secure RCP 2B when the CBO went to zero gpm, no additional pump assembly damage occurred and an uncomplicated, safe shutdown of the plant was accomplished. However, the potential worst case implications of a cracked rotating baffle have been reviewed to ensure a safe shutdown of the plant would still have occurred, even without prompt operator action.

Because a RCP is not credited for accident mitigation or safe shutdown, the unavailability of a RCP is not safety significant. Loss of flow from a single RCP coastdown during full power operation is analyze in FSAR Section

15.3.1.1 with acceptable results. However, two other unlikely events can be postulate to be potential results of a cracked rotating baffle. The first is a perforation in the seal cooling heat exchanger caused by debris or unbalance of the cracked rotating baffle. The second is a seized RCP shaft, caused by the unbalanced rotating baffle being wedged into the low tolerance space between the stationary heat exchanger cylinders. The consequences of a seized RCP shaft are analyzed in FSAR Section 15.3.3.1. The likelihood of a seized RCP shaft has been determined to be negligible because the bolts attaching the baffle to the shaft would shear before shaft seizure. Therefore, this potential failure is not discussed further.

The aspect of rotating baffle damage that is reviewed for safety significance is the potential effect on the RCP Seal Cooler, which is cooled by CCW. In this instance, the rotating baffle damage was not sufficient to perforate the wall of the cooler. However, if the rotating baffle was damaged such that it breached the cooler wall, a path from the RCS to outside containment through the CCW system would exist, thereby causing an Interfacing System Loss of Coolant Accident (ISLOCA). Because an ISLOCA allows RCS fluid to leak outside of containment [NH], no fluid collects in the Safety Injection [BQ] sump for recirculation after depletion of the Refueling Water Storage Pool (RWSP). Therefore, this event is more severe than an in-containment LOCA because recirculation cannot occur.

A previous engineering evaluation reviewed the potential of heat exchanger failure in response to IN 8954, "Potential Overpressurization of the Component Cooling Water System". This evaluation reviewed the heat exchanger from hydrostatic, hydrodynamic, and thermal stress perspectives, and concluded that cracking of the heat exchanger, causing a break in the RCS pressure boundary, was not a credible event. However, the evaluation did not account for potential damage caused by a cracked rotating baffle. Previous occurrences of rotating baffle (bolting) failures causing forced shutdowns have

occurred at Waterford 3. These occurrences resulted in slight heat exchanger damage in the past, but did not result in heat exchanger perforation.

In the event of a perforated heat exchanger, the leakage would be mitigated by automatic closure of RCP Seal Cooler Isolation Valves CC-666A&B, CC-6651A&B, CC-679A&B, and CC-680A&B, which are located inside containment at the inlet and outlet of the seal coolers. These valves are automatically closed when the CCW outlet temperature at the heat exchanger reaches 155 degrees F. Normal CCW temperature at the outlet of the heat exchanger is approximately 130 degrees F. Any significant leak of RCS fluid at 545 degrees F into the CCW side of the heat exchanger is expected to cause the temperature to increase above 155 degrees F. Prior to valve closure, an alarm annunciates on CP-2 at 145 degrees F outlet temperature. The valves close at 155 degrees F, but manual reset is allowed. If the valves are reset and temperature does not decrease below 145 degrees F within 100 seconds, the valves will reclose. This function is designed to detect a cooler leak/break and isolate the affected cooler, and if manually reset will reisolate the cooler, making the operators aware of the potential ISLOCA at the RCP cooler. Prompt operator action will also be facilitated through a radiation monitor located on the RCP-CCW return header, with annunciation provided on CP-2. In addition, each CCW loop contains a radiation monitor which should indicate rising trends on CP-6 and/or alarms on CP-8.

The isolation valves are 1500 lb., flow under the seat, air operated globe valves. Upon review of the draft design basis review calculation for these valves, it is concluded that the valves are capable of closing at RCS pressures. Therefore, if the heat exchanger were to be perforated by the rotating baffle failure, the potential ISLOCA would be quickly isolated through automatic action of these valves.

The piping between the coolers and the isolation valves is described on the applicable isometric drawings as being designed for 175 degrees F and 125 psig, and was hydrostatically pressure tested to 156 psig. However, per the isometric drawings, this piping is Schedule 80, ANSI-106, Grade B, carbon

steel. Per the National Valve and Manufacturing Company, the maximum working pressure of this type of piping is 2983 psig at 650 degrees F.

Connection flanges between this piping and the heat exchanger are classified at 1500 lb. class. Per Mark's Handbook, a Class 1500, A105, carbon steel flange is rated for 2685 psig at 650 degrees F. Therefore, it is unlikely that the flange, piping or valves would fail; and the breach would be isolated. However, even if this piping or flange were to fail, the result would be bounded by the small break LOCA analysis described in FSAR Section 15.6.3.

If further failures of the cooler isolation valves to close or a loss of offsite power (LOOP) were postulated, an ISLOCA could occur with a path for RCS fluid outside of containment. A LOOP causes a loss of electrical supply to the Instrument Air (IA) [LID] compressors [CMP], thus causing a loss of IA. Because the CCW isolation valves [ISV] are fail-open air-operated valves (AOVs), these valves would open, once IA is lost. However, it is estimated that it would take more than 8 hours to exhaust the RWSP water supply for the postulated ISLOCA. This allows sufficient time for operators to take recovery actions, such as loading an IA compressor onto an EDG, thereby restoring air to the seal cooler isolation valves and isolating the leak. The probability that this scenario (catastrophic failure of the rotating baffle, perforation of the heat exchanger, LOOP, and failure of recovery actions) could lead to core damage has been calculated as 1.5E-6.

Assuming that the ISLOCA is not isolated, overpressurization of the CCW system could occur. Upon Safety Injection Actuation Signal (SIAS) initiation, the two CCW trains split into redundant A and B trains, with the A train continuing to supply the RCP coolers. However, the two trains continue to be connected through their common surge tank. Therefore, although the RCS fluid will directly flow into the A train, causing potential overpressurization, affects will also be seen by the B train once the surge tank is filled and pressurized. The overall affects of the overpressurization on CCW operation should be small. Although some decreased efficiency will be seen due to the influx of the higher temperature RCS

fluid into the CCW system, the impact should be minimal and not affect the components which CCW supports. The small increase in system pressure should also not affect pump operation. The largest impact of the CCW overpressurization is the potential for flooding in essential areas due to overfill of the surge tank and lifting of relief valves. The potential overpressurization of the CCW system due to uncontrolled make-up has been previously reviewed for worst case scenarios in a previous corrective action document operability evaluation. This corrective action document considered the flooding effects of fluid in the minus 35 elevation level,

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primarily due to overflow of the CCW surge tank into the waste tanks, and from floor drain collection. This evaluation bounds the effects of the postulated ISLOCA and concluded that no additional safety-related equipment will be affected. Further, because the areas of potential flooding will not affect any of the operator actions postulated in the RCP/ISLOCA scenario, the potential overpressurization and overfilling of the CCW system will not increase the probability of core damage previously mentioned.

SIMILAR EVENTS

A search of Waterford 3 and industry events was conducted. The search did not reveal any other failures of rotating baffles where the baffle was cracked or otherwise failed. However, two previous corrective action documents were written regarding rotating baffle problems. CR-1995-0536 documents the failure of six baffle bolts on RCP 2B. CR-1996-1048 documents that rubbing contact with the baffle damaged the heat exchanger.

ADDITIONAL INFORMATION

Energy Industry Identification System (EIIS) codes are identified in the text within brackets

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Attachment "COMMITMENT IDENTIFICATION/VOLUNTARY ENHANCEMENT FORM" omitted.

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